10 CFR 50.73

Boston Edison
Pilgrim Nuclear Power Station
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U.S. Nuclear Regulatory Commission
Attn.: Document Control Desk
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Docket No. 50-293
License No. DPR-35

The enclosed Licensee Event Report (LER) 97-026-00,"Automatic Scram Due to High Reactor Water Level During Power Ascension", is submitted in accordance with 10 CFR Part 50.73.

This letter contains no commitments.

Please do not hesitate to contact me if there are any questions regarding this report.

NLDesmond

N.L. Desmond

Enclosure: LER 97-026-00
KRD/dcg/9702600

IE22/1

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RECIP.NAME	RECIPIENT AFFILIATION

SUBJECT: LER 97-026-00:on 971206,automatic scram was initiated.Caused by high reactor water level during power ascension.Pilot valve clip was repositioned inside Bailey positioner & valve was stroked smoothly.W/980105 ltr.

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NRC Form 366 (4-95)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

LICENSEE EVENT REPORT (LER)

(See reverse for number of digits/characters for each block)

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TITLE (4)
Automatic Scram Due to High Reactor Water Level During Power Ascension

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME: N/A	DOCKET NUMBER: 05000
12	06	97	97	026	00	01	05	98	FACILITY NAME: N/A	DOCKET NUMBER: 05000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (Check one or more) (11)

OPERATING MODE (9)	Y		20.2201 (b)		20.2203(a)(2)(v)		50.73(a)(2)(i)(A)		50.73(a)(2)(viii)
POWER LEVEL (10)	75		22.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)	X	50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Kristin R. DiCroce - Senior Regulatory Affairs Engineer	(508) 830-7667

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
A	SJ	FCV	B042	Y					

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE(15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On December 6, 1997, at 0907 hours, an automatic scram occurred at approximately 75 percent reactor power. The scram was initiated as a result of a turbine trip due to high reactor water level (>45 inches) experienced during power ascension following an outage. The turbine trip/reactor scram resulted in the insertion of the control rods and transfer of power to the startup transformer.

The cause of the scram was the failure of the "A" feedwater regulating valve (FRV) in the full open position due to the misalignment of the valve clip inside the pilot valve assembly of the positioner. Corrective action taken included replacing the valve positioner and performing calibration adjustments.

The scram occurred while the reactor mode selector switch was in the RUN position. The reactor vessel pressure was approximately 1004 psig with the reactor water temperature at the saturation temperature for the reactor pressure. The event posed no threat to public health and safety.

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BACKGROUND

A technical specification required shutdown was completed on November 23, 1997, because two main steam isolation valves (MSIVs) in separate main steam lines were inoperable (LER 97-025-00). During power ascension from the MSIV forced outage, and while at 75 percent power, the "A" feedwater regulating valve (FRV) failed in the full open position causing an increase in reactor water level.

The status of the systems just prior to the event was as follows:

- Reactor power was approximately 75 percent with the reactor core flow rate at approximately 38.4 E+06 pounds per hour.

- The reactor vessel pressure was approximately 1004 psig.

- The reactor water level was approximately +30 inches (narrow range level). All three reactor feed pumps (RFPs), "A," "B," and "C," were in service. Feedwater flow to the reactor vessel was approximately 5.7 E+06 pounds per hour. The feedwater level control system was in service in the three element automatic control mode (reactor water level, feedwater flow, and steam flow). The main steam flow rate was approximately 5.5 E+06 pounds per hour.

- The 4.16 Kv auxiliary power distribution system buses A1 through A6 were being powered from the main generator via the unit auxiliary transformer with the fast transfer control switches in the ON position. The startup transformer was in standby service. The 345 Kv transmission lines (342 and 355) were energized. The 345 Kv switchyard ring bus was energized with the switchyard air circuit breakers 102, 103, 104, and 105 in the closed position. The emergency diesel generators ("A" and "B") were in standby service. The shutdown transformer, the station blackout diesel generator, and related bus (A8) were in standby service.

EVENT DESCRIPTION

On December 6, 1997, at 0907 hours, an automatic reactor protection system (RPS) scram signal and scram occurred while at approximately 75 percent reactor power. The scram was the result of an automatic turbine trip caused by high reactor vessel water level. The failing open of feedwater regulating valve, FV-642A, was the cause of the increased reactor vessel water level.

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At approximately 0907 hours, the operating crew was first alerted to the condition by the reactor vessel water level high alarm which annunciated at approximately +32 inches. The feedwater flow indication in the "A" feedwater line was also observed to be pegged upscale indicating a failure of the "A" FRV. The automatic feedwater level control system (FWLCS) responded to the increase in feedwater flow and increasing vessel level by generating a closure signal to both feedwater valves. The "B" FRV closed as expected, but the "A" FRV remained fully open. The resultant reduction in feedwater flow through the "B" FRV was insufficient to overcome the increase in feedwater flow through the "A" FRV due to that valve having failed full open.

Additionally, the operating crew placed the "A" feedwater level controller in manual and attempted to close the valve manually from the control room. This action also proved ineffective due to the mechanical nature of the failure. Reactor water level continued to rise, and as water level approached the high level turbine trip setpoint (approximately +45 inches), the operating crew prepared to initiate a manual reactor scram. Prior to manual scram initiation, an automatic main turbine trip occurred when water level reached the high water level turbine trip setting. The turbine trip generated an automatic reactor scram on main turbine stop valve closure and control valve fast closure.

The scram signal resulted in the automatic insertion of the control rods, and the automatic transfer of buses A1 through A6 from the unit auxiliary transformer to the startup transformer.

Initial control room licensed operator actions were taken in accordance with procedure 2.1.6, "Reactor Scram." The actions included moving the reactor mode selector switch to the SHUTDOWN position.

Following the scram, the reactor vessel water level decreased as expected. The decrease to approximately +3.2 inches (narrow range) was due to the combined effects of the decrease in the reactor water void fraction resulting from the scram and the decrease in steam flow to the main condenser. The decrease to less than the low reactor water level setpoint (calibrated at approximately +12 inches) resulted in the automatic initiation of the primary containment isolation control system (PCIS) and reactor building isolation control system (RBIS) as designed.

The PCIS initiation resulted in the following expected designed responses:

- The PCIS Group 2 (drywell) isolation valves automatically closed.
- The PCIS Group 3 residual heat removal (RHR) system shutdown cooling suction piping isolation valves, MO-1001-47 and -50, remained closed. The RHR system low pressure coolant injection loop "A" valve, MO-1001-29A, and loop "B" valve, MO-1001-29B, remained closed.

- The PCIS Group 6 reactor water cleanup (RWCU) isolation valves closed automatically. The Group 6 circuitry was subsequently reset, and the RWCU system was put into service in the reject mode.

The RBIS initiation resulted in the automatic start of the standby gas treatment system (SGTS) trains "A" and "B" and automatic closing of the secondary containment ventilation supply and exhaust dampers.

Continuing control room operator response included activities for verifying the insertion of the control rods. Six control rods did not initially indicate a full-in position immediately following the scram. Emergency Operating Procedure (EOP), EOP-02, "Reactor Pressure Vessel (RPV) Control - Failure to Scram," and procedure 5.3.23, "Alternate Rod Insertion," were entered to verify proper insertion. Procedure 5.3.23 required closing the control rod drive (CRD) system charging water valve, HO-301-25. After this valve was closed, the six rods indicated proper insertion, and EOP-02 was exited. All control rods were determined to be inserted at approximately 15 minutes after the scram, and EOP-02 was exited.

At 0918 hours, the RWCU system isolated due to a sensed high temperature at the outlet of the RWCU non-regenerative heat exchanger. The temperature was sensed by temperature element, TE-1291-12A/TIS-1291-13A. This isolation function is nonsafety-related. The circuit was reset, and the RWCU system . was returned to service in the reject mode.

The NRC Operations Center was notified of the event at 1017 hours in accordance with 10CFR50.72(b)(2)(ii). The report was made because of the actuation of the reactor protection system (RPS) due to a turbine generator trip on high reactor water level (>45 inches) and the actuation of PCIS Groups 2 and 6 on low reactor water level after the reactor scram.

At 0205 hours on December 7, 1997, the RHR system was put into service in the shutdown cooling mode. Cold shutdown conditions were achieved when the reactor vessel water temperature was reduced to less than 212 degrees Fahrenheit.

A post trip review was conducted in accordance with procedure 1.3.37, "Post Trip Review." A critique of the event was also conducted. The post trip review and critique included applicable personnel including the operators on shift at the time of the event.

Problem reports were written to document the scram and other observations prior to, during, or after the event and included the following:

- PR 97.9765 was written to document the turbine trip/reactor scram.
- PR 97.9766 was written to document intermediate range monitor (IRM) "D" spiked high at the time of the scram.
- PR 97.9768 was written to document that the MSIV logic light extinguished on the "A" side.
- PR 97.9767 was written to document that IRM "A" spiked causing a RPS channel "A" trip signal after the reactor was shutdown.

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- PR 97.9769 was written to document that no scram time data was obtained for control rod 38-31 during the reactor scram.
- PR 97.9771 was written to document that point #1 (reactor vessel flange) on temperature recorder, TR-263-104, was tracking erratically during plant cooldown (LER 97-025-00).
- PR 97.3583 was written to document the problem with the position indication of the six control rods.

CAUSE

The RPS scram signal and scram was the result of a turbine trip due to a high reactor water level condition.

The high water level condition was caused by the opening of the "A" FRV (FV-642A). The opening of FV-642A was caused by the inadvertent misalignment of the pilot valve clip inside the pilot valve assembly of the Bailey positioner for the "A" FRV. The pilot valve clip was misaligned when the positioner was opened during the MSIV forced outage for troubleshooting. After the scram, when the positioner for the "A" FRV was opened, the pilot valve clip was found to be displaced. This caused the positioner's pilot valve stem to move downwards, porting air from the top of the "A" FRV actuator, allowing the valve to go full open.

The feedwater regulating valve, FV-642A, is a Copes-Vulcan fourteen inch, 900 psi, double poppet, balanced, hydraulically dampened, diaphragm operated control valve equipped with a D100-160 actuator and a Bailey AV1 positioner.

CORRECTIVE ACTION

The following corrective action was taken for FV-642A:

- The pilot valve clip was repositioned inside the Bailey positioner, and the valve was stroked smoothly. The calibrations were satisfactorily completed.

- Subsequently, the valve positioner was replaced after the troubleshooting was completed. Calibration adjustments and subsequent loop calibration checks were performed as prescribed by MR19702989. Post work testing included an overall system response check. This was conducted by manually dialing a setpoint change into the master controller to simulate changing water level conditions while observing the "A" and `B' feedwater control valve responses. A low water level was simulated and both valves were observed to move in the open direction. A high water level was then simulated and both valves were observed to move in the close direction. The valves were observed to move smoothly and in unison in the appropriate directions.

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- Procedure 3.M.2-10, attachment 4, "Feedwater Control Valve Isolation and Maintenance " was performed satisfactorily prior to startup.

The `B' FRV was tested in accordance with 3.M.2-10, "Feedwater Control Valve Isolation and Maintenance," and no problems were observed.

As part of our normal training process, this event will be reviewed for inclusion in applicable plant status updates.

Corrective actions for the other problem reports mentioned in this report were or are being dispositioned via the problem report corrective action process.

The "all rods full-in" issue has been previously recognized as a significant operator workaround for which corrective actions are being implemented. This issue was the subject of VIO 97-01-02, and its corrective actions are being tracked via problem report PR 96.9206.

SAFETY CONSEQUENCES

This event posed no threat to the public health and safety.

The RPS scram signal was the designed response to the turbine trip.

The decrease in the reactor water level immediately after the scram was the expected response to the reactor water void fraction decrease (shrink) resulting from the scram and the decrease in steam flow to the main condenser. The resultant PCIS and RBIS actuations immediately after the scram were the expected designed responses to a low reactor vessel water level condition (approximately +12 inches).

The technical specification table 3.2.B trip setting for automatic actuation of the core standby cooling systems (CSCS) is approximately -46.3 inches. During the event, the lowest reactor vessel water level that occurred, +3.2 inches, was approximately 49.5 inches above the CSCS setpoint. In addition, the level was approximately 130 inches above the level (-127 inches) that corresponds to the top of the active fuel zone.

The lowest reactor water level that occurred, approximately +3.2 inches, was also greater than the setpoint, calibrated at approximately -46.3 inches, that initiates the ATWS system functions for a recirculation pump trip and alternate rod insertion.

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This report is submitted in accordance with 10 CFR 50.73(a)(2)(iv) because the actuation of the RPS was not planned.

This report is also submitted in accordance with 10 CFR 50.73(a)(2)(iv) because PCIS Group 2 and Group 6 isolations, although designed responses to a low reactor vessel water level condition, were not planned.

SIMILARITY TO PREVIOUS EVENTS

A review of Pilgrim Station LERs submitted since 1984 was conducted. The review focused on LERs submitted in accordance with 10 CFR 50.73(a)(2)(iv) that involved a misalignment of a valve clip inside the pilot valve assembly of the positioner. The review identified reactor vessel water level related scrams reported in LERs. None involved a misaligned valve clip inside the pilot valve assembly of the positioner of the feedwater regulating valve, FV-642A.

A review was also conducted of other Pilgrim Station events that involved a misalignment of the valve clip inside the pilot valve assembly of the positioner. The review identified no previous similar events with FV-642A.

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for this report are as follows:

COMPONENTS	CODES
Rod (control rods)	ROD
Valve, control, flow (FV-642A/B)	FCV

SYSTEMS	
Condensate system	SD
Containment isolation control system (PCIS)	JM
Control rod drive system	AA
Engineered safety features actuation system (RPS, PCIS)	JE
Feedwater level control system	JB
Feedwater system	SJ
Incore monitoring system (neutron monitoring system)	IG
Main turbine system	TA
Plant protection system (RPS)	JC